OPPD

Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247
402/636-2000

July 30, 1991
LIC-91-177R

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

Gentlemen:

SUBJECT: Fort Calhoun Station (FCS) Post Accident Sampling System (PASS) Reactor Coolant (RC) Dissolved Gas Analysis Sequence

Omaha Public Power District (OPPD) proposes to revise commitments concerning post-accident RC dissolved gas analysis using the FCS PASS. A list of reference documents related to this issue is contained in Appendix A. References 1-4 discuss previous OPPD commitments concerning the ability of the PASS to meet NUREG-0737 Item II.B.3 criteria. Reference 5 discusses FCS PASS compliance with Regulatory Guide 1.97, Revision 2 criteria. Recent submittals (References 6-8) have discussed OPPD experiences in attempting to resolve difficulties in the PASS RC automated dissolved gas analysis sequence. OPPD stated in Reference 8 that the architect/engineer and PASS supplier would perform an evaluation of the PASS RC dissolved gas analysis sequence. This evaluation was completed in May 1991, and OPPD recently completed its review of the findings.

OPPD has concluded that the reliability of the PASS automated dissolved gas analysis sequence, especially under accident conditions, is difficult to assure. Moreover, the value of the analysis results for accident mitigation is limited and redundant to other available information. Therefore, OPPD proposes to delete credit for the PASS automated dissolved gas analysis sequence and retain only the ability to obtain an undiluted pressurized RC grab sample for offsite dissolved gas analysis following an accident. Appendix B contains the details of this proposal, which meets the intent of NUREG 0737 Item II.B.3 and will not adversely affect public health and safety.

OPPD requests NRC concurrence with this proposal based on the enclosed Appendix B justification. This justification includes the precedent set by the NRC acceptance of a similar proposal from Baltimore Gas and Electric Company for the Calvert Cliffs Nuclear Power Plant (Docket Nos. 50-317 and 50-318), as documented in the Safety Evaluation Report dated May 6, 1986 (Reference 9).

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If you should have any questions, please contact me.

Sincerely,

W. G. Gates

W. G. Gates
Division Manager
Nuclear Operations

WGG/sel

Enclosures

c: LeBoeuf, Lamb, Leiby & MacRae
R. D. Martin, NRC Regional Administrator, Region IV
W. C. Walker, NRC Project Manager
R. P. Mullikin, NRC Senior Resident Inspector

REFERENCES

1. Letter from OPPD (W. C. Jones) to NRC (R. A. Clark) dated December 3, 1982 (LIC-82-389)
2. Letter from NRC (R. A. Clark) to OPPD (W. C. Jones) dated January 31, 1983
3. Letter from OPPD (W. C. Jones) to NRC (R. A. Clark) dated April 15, 1983 (LIC-83-083)
4. Letter from NRC (C. M. Trammell) to OPPD (W. C. Jones) dated September 1, 1983
5. Letter from OPPD (R. L. Andrews) to NRC (D. E. Sells) dated October 21, 1986 (LIC-86-532)
6. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated May 16, 1990 (LIC-90-0247)
7. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated January 7, 1991 (LIC-90-1034)
8. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated March 22, 1991 (LIC-91-042R)
9. Letter from NRC (D. H. Jaffe) to Baltimore Gas & Electric Company (J. A. Tiernan) dated May 6, 1986 (Docket Nos. 50-317 and 50-318)

OMAHA PUBLIC POWER DISTRICT PROPOSAL TO REVISE NUREG-0737 ITEM II.B.3 COMMITMENTS

1.0 Purpose

Omaha Public Power District (OPPD) proposes to revise previous commitments to the NRC relative to Fort Calhoun Station (FCS) Unit No. 1 compliance with the requirements of NUREG-0737 Item II.B.3. For post-accident dissolved gas analysis of Reactor Coolant (RC) samples, OPPD proposes to delete credit for the PASS automated dissolved gas analysis sequence and retain only the ability to obtain an undiluted pressurized RC grab sample for offsite dissolved gas analysis

2.0 Background

Listed below are the regulatory bases for post-accident dissolved gas sampling (NUREG-0737, and RG 1.97, Rev. 2) along with current OPPD commitments. (Note: referenced documents are listed in Appendix A of LIC-91-177R.)

Summary of Applicable NUREG-0737 Item II.B.3 Requirements for Dissolved Gas Analysis

Criterion (1):
The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

Criterion (2) (c):
The licensee shall establish an onsite radiological and chemical analysis capability to provide, within three-hour time frame established above, quantification of ... dissolved gases (e.g., hydrogen).

Criterion (4):
Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or hydrogen gas in reactor coolant samples is considered adequate. Measuring the oxygen concentration is recommended, but is not mandatory.

Criterion (6):
The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19.

Criterion (8):
If inline monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable.

Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident, and at least one sample per week until the accident condition no longer exists.

Criterion (10):
Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

Criterion (11):
In the design of the post accident sampling and analysis capability, consideration should be given to the following items:

(a) Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the Reactor Coolant System (RCS) or containment, for appropriate disposal of the samples, and for flow restriction to limit reactor coolant loss from a rupture of the sample line. The post-accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

(b) The ventilation exhaust from the sampling station should be filtered with charcoal absorbers and high efficiency particulate air (HEPA) filters.

The criteria were developed to evaluate the RCS and Containment Sump conditions and provide information on the degree of core damage within a three hour time frame from the time a decision is made to take a sample. Concerns were raised that during an accident the potential exists for generation of gases which are indicative of fuel cladding damage (dissolved gases such as hydrogen). NUREG-0737 also requires the capability of monitoring RCS hydrogen levels up to 30 days after an accident. This requirement is for long term RCS integrity monitoring. Dissolved gas monitoring was also intended to provide information to allow prediction of void formation for anticipated plant conditions which could have a disruptive influence on core cooling (Plant Recovery Phases).

<u>Regulatory Guide (RG) 1.97, Rev. 2 Criteria</u>

RG 1.97, Rev. 2 required that RCS/Sump samples be monitored for hydrogen or total gas. This variable was categorized as Type E, Category 3. Type E variables are those variables which determine the magnitude of a release. A Type E variable provides the operator with information to evaluate effectiveness of actions. The variable types are broken down into "subtypes" or Categories. The dissolved gas monitoring variable was determined to be a Category 3 subtype, which means this variable is a backup to a key variable.

Summary of Previous OPPD Commitments to Satisfy NUREG-0737 and RG 1.97, Rev. 2 Requirements for RCS Dissolved Gas Analysis

The Fort Calhoun Station PASS was developed and implemented to meet the criteria of NUREG-0737 Item II.B.3. The only on-site method for analyzing RC dissolved gas quantity following an accident uses the PASS instrumentation. The PASS has an automatic dissolved gas analysis system which can be operated remotely from panel AI-194 located in the FCS hot lab. The process for measuring dissolved gases is based on sample depressurization, and on the assumption that all the gas present in the RCS following a design basis accident (DBA) is hydrogen. The system operates by trapping a known volume of reactor coolant in a high pressure sample flask. The sample is cooled, then expanded into a second flask of known volume. Analytical equations are then provided to derive the amount of gas evolved and still in solution using Operating Instruction OI-SL-2.

Provisions for grab sampling of the reactor coolant, containment sump and containment atmosphere samples exist within the PASS. Grab samples can be taken of pressurized or unpressurized samples and of diluted or undiluted samples. However, as previously stated in References 1 and 3, OPPD does not have the ability to perform onsite analysis of undiluted RC grab samples. In addition, no arrangements have been made for offsite analysis of high pressure, undiluted samples.

In Reference 4, the NRC approved the FCS PASS method for RC dissolved gas analysis. In Reference 5, it was shown that the PASS as originally designed meets RG 1.97, Rev. 2 requirements.

3.0 Fort Calhoun Station PASS Dissolved Gas Sequence Operational Concerns

The dissolved gas analyzer sequence has been subject to numerous problems since being declared operational in 1983, due to the complexity and sensitivity of the remote, automatic system. Dissolved gas samples obtained from the PASS under normal RCS conditions have yielded results outside of recommended tolerances when compared against results from daily RCS grab samples. Several factors have been found to be responsible for the out of tolerance conditions. These include: (1) inadequate sample cooling, (2) leakage through the Autoclave isolation valves, (3) incorrect equations modelling the gas expansion, (4) inadequate flushing with nitrogen and demineralized water, and (5) instrument accuracy. Two of these factors, valve leakage and instrument accuracy, have not been resolved, as noted below. In addition, the PASS system requires extensive maintenance activity to maximize availability.

A leak tight system is essential to insure the reliability of the expansion method for determination of dissolved gases in the RCS. The dissolved gas analysis sequence depends upon complete isolation of at least twenty-two valves during the analytical steps in the sequence. In an accident condition with core damage, the potential exists for crud entrapment between the seats of these valves resulting in leakage. This leakage could cause the dissolved gases to come out of solution prematurely, resulting in a nonconservative calculation of dissolved gas quantity. In May 1991, the designer of the PASS conducted an evaluation and stated that "an assumption that the system is leak tight is essential to obtain meaningful results. Any leakage would adversely affect the function of the analyzer."

The dissolved gas determination is highly dependent upon instrumentation accuracy. Instrumentation error in pressure and temperature measurement can offset the accuracy of gas measurement by several orders of magnitude. The sample analysis is dependent upon output from two pressure transmitters and two temperature elements. These instruments are calibrated annually in accordance with information submitted in OPPD's response (Reference 3) to open items identified in the draft SER (Reference 2). The initial pressure readings are obtained from a transmitter with an accuracy of +/-50 psi (range 0-2500 psig). The final pressure of the expanded samples is read from a transmitter with an accuracy of +/-0.2 psi (range 0-10 psig). For low initial RCS pressures (e.g., a large break LOCA), the effect of the error will be greater than during normal operating modes with a RCS pressure of 2100 psig. The designer of the PASS has also identified that the range of the final pressure transmitter could be an issue if the actual pressure was less than atmospheric. This could occur if the ambient pressure increased after a sample was initially isolated. In addition, if the expanded sample did not evolve enough dissolved gas to overcome the shrinkage of the liquid sample, this would adversely affect the ability to measure samples with small quantities of gas.

Due to system complexity, maintenance is an additional problem. The PASS RC automated dissolved gas analysis sequence is interfaced with the nitrogen and demineralized water flush systems. The sequence relies on instrument air for positioning thirty-four air operated isolation valves. The sequence is interfaced with the containment atmosphere for waste gas return; liquid waste is returned to either the Volume Control Tank or Waste Disposal system. The sample lines of the dissolved gas analysis sequence are several hundred feet in length. The sequence must rely on liquid and gas drain pumps which return RCS fluids to containment or radioactive fluid handling systems. Problems have arisen due to failure of level controllers in drain tanks causing pumps to not maintain system levels. Drain lines were originally inappropriately sized to handle liquid wastes. Valve failures have occurred on a regular basis since the dissolved gas sequence was declared operational. Typical valve failures include leaking diaphragms in instrument air valves and electrical burnout of solenoid-operated water bath valves.

4.0 Contribution of Dissolved Gas Analysis to Core Damage Assessment

Results from the dissolved gas analysis were intended to assist personnel performing assessment of fuel failure. However, it can take up to an hour to complete a PASS dissolved gas analysis as dictated by OPPD procedure OI-SL-2. Sampling the RCS for total dissolved gases through the PASS will not provide indications in real time of core uncovering and the subsequent fuel cladding damage. Information obtained via dissolved gas sample analysis is only confirmatory with regard to the possible extent of core damage. The PASS is not capable of supporting real time decisions made during the early stages of accident progression or recovery except for an extremely slow transient. An analyst could not make decisive conclusions about the extent of fuel damage based solely on RCS dissolved hydrogen levels, since there are other sources of hydrogen generation specific to FCS. If the RCS were breached (large break LOCA), reactor coolant would be exposed to containment atmospheric conditions and containment spray would be expected to operate. Under these conditions, any hydrogen generated from fuel cladding failure (i.e., Zirconium-water reaction) would be masked by hydrogen generated from the interaction of containment spray with aluminum and zinc materials inside containment.

Depressurized reactor coolant in contact with a containment environment rich in hydrogen could absorb gaseous hydrogen to saturation level, which could significantly skew dissolved hydrogen level results.

Neither FCS Emergency Operating Procedures (EOPs) or the Combustion Engineering Emergency Procedures Guidelines (upon which the EOPs are based) reference drawing a PASS reactor coolant sample for dissolved gas analysis immediately following an accident in order to assess fuel cladding failure. Safety related, real time instrumentation to monitor fuel failure without input from the PASS is available to the operators and Technical Support Center (TSC) personnel. This instrumentation includes the Core Exit Thermocouples (CETs)/Heated Junction Thermocouples (HJTCs) and Containment Radiation and Hydrogen Monitors. RCS hydrogen concentration can be correlated from containment atmosphere hydrogen monitors for large break scenarios, and total gas can be correlated from subcooled margin monitors, plant pressure, and temperature readings.

The potential for non-condensible gas voids interfering with core cooling is eliminated by the reactor vessel head vent system. EOP-3, Rev. 15.1, Part 1 provides guidance for identifying and dealing with void formations. As outlined in the EOPs, voiding in the RCS may be shown by any of the following indications, parameter changes, or trends: pressurizer level increasing, reactor vessel level monitoring, and erratic steam generator Delta p. By monitoring these real time indicators, a status of coolant heat transfer effectiveness can be extrapolated.

FCS Emergency Plan Implementing Procedure EPIP-TSC-8 provides guidance on performing core damage assessment. PASS operating procedures are listed as reference materials in EPIP-TSC-8. The emergency plan procedure provides guidance for core uncovering prediction and verification. The EPIPs require monitoring of the CETs, pressurizer pressure and level, and core level indications. An estimation of core damage can be provided through running a PASS isotopic analysis; however, EPIP-TSC-8 notes that samples obtained during rapidly changing plant conditions should not be weighed heavily into the assessment of core damage. The results of the isotopic analysis are recorded on a worksheet along with reactor vessel level, pressurizer level and containment sump level. The computer program "UTYPASS" is used to make an estimation of core damage from isotopic inputs. Dissolved gas quantification is not referenced in this EPIP based on the need for real time information. Thus, short term determination of dissolved gas quantity does not serve the original regulatory intent, which is to enable operators or analysts to make core damage assessments useful in accident mitigation.

The negligible value of dissolved gas analysis in accident mitigation has also been confirmed by the NRC. NUREG/CR-4330, Volume 3, Appendix A notes the following points:

> During accident management the PASS should only be relied on for boron analysis. Atmospheric containment hydrogen analyzers are considered useful in responding to a transient in the accident management phase. During this phase, dissolved hydrogen analysis would not be used to terminate or limit the progression of a serious accident. In the event of an accident which progresses to core melt in less than three hours, other indicators such as high-range containment monitors and core exit thermocouples should be used to classify an emergency versus using PASS.

NUREG-4330 also provides guidance on core damage assessment which agrees that the CETs, HJTCs, containment hydrogen, and radiation monitor instrumentation provide real time direct readouts in the control room and TSC (via the Emergency Response Facilities Computer System) which are immediately and continuously available to decision makers.

5.0 Regulatory Precedence

Precedent for NRC acceptance of this proposal was provided by previous NRC approval of a similar proposal by Baltimore Gas and Electric Company. As noted in the May 6, 1986 Safety Evaluation Report (SER) for Calvert Cliffs Nuclear Power Plant (Reference 9), the NRC agreed with modification of the original Calvert Cliffs PASS to rely on grab sample analyses versus inline sampling methods. This was based on maintenance and reliability problems, as well as instrumentation inaccuracies associated with the inline system. These are the same problems noted for the FCS PASS system.

6.0 Conclusion

This proposed revision of NUREG-0737 Item II.B.3 commitments for FCS meets the original intent of NUREG-0737 and Regulatory Guide 1.97, Rev. 2. An evaluation of emergency procedure guidelines, specific accident scenarios involving potential core damage, and later regulatory guidance revealed that analyzing RC samples after an accident serves only to complement and confirm other, more valuable sources of information. These other sources, which are referenced in EOPs and EPIPs, will allow a timely determination to be made in the Technical Support Center and the Emergency Operations Facility regarding the degree of core damage. For long term post-accident recovery, other means of information are more pertinent than the PASS dissolved gas analysis. Following an accident, the Site Director would decide if drawing an undiluted grab sample for dissolved gas analysis was warranted, considering the limited additional value for accident mitigation versus the potential for radiation dose associated with the sampling process. Subsequent to any decision to draw an undiluted sample, arrangements would be made for offsite analysis.

Except for deleting the automated dissolved gas analysis sequence, other functions (such as inline isotopic, pH, Boron and Chloride analyses) associated with the PASS will be maintained within the recommended tolerances.

OPPD concludes that deletion of reliance on the PASS automated dissolved gas analysis sequence still meets the intent of NUREG 0737 Item II.B.3 and will not adversely affect public health and safety.